СПИСОК ЛИТЕРАТУРЫ

1. Хвостиков А.С., Богданов К.С. Модернизация башенных градирен эксплуатируемых в сложных климатических условиях // Энергосбережение и водоподготовка. – 2019. – С. 22–26. – URL: https://elibrary.ru/item.asp?id=41262938 (дата обращения: 25.10.2023). – Режим доступа: Научная электронная библиотека eLIBRARY.RU. – Текст : электронный.
2. Калатузов В.А. Низкопотенциальная часть тепловых электростанций одна из причин ограничения их мощности // Энергосбережение и водоподготовка. – 2010. – С. 34–37. – URL: https://elibrary.ru/item.asp?id=15223315 (дата обращения: 20.10.2023). – Режим доступа: Научная электронная библиотека eLIBRARY.RU. – Текст : электронный.
3. Сибирская генерирующая компания : официальный сайт – URL: https://sibgenco.ru/ (дата обращения: 26.10.2023).
4. Лаптев А.Г., Ведьгаева И.А. Устройство и расчет промышленных градирен: монография. – Казань: КГЭУ, 2004. – 180 с.
5. Лаптева Е.А., Лаптев А.Г., Фарахов М.И. Показатели энергоэффективности градирен // Надежность и безопасность энергетики. – 2018 – Т. 11, № 3. – С. 217–221. https://doi.org/10.24223/1999-5555-2018-11-3-217-221

EFFECT OF POWER EXCURSION ON THE THERMAL CHARACTERISTICS OF VVER-1200 REACTOR

C.J. Odii, A.G. Korotkikh

Tomsk Polytechnic University,
SEPE, SNSE, Group A2-43I

Scientific supervisor: A. G. Korotkikh, DSc, Professor of BRC SEPE TPU

This work involved performing a thermal hydraulics calculation on VVER-1200 reactor. In this calculation, the thermal power of the reactor was varied in steps from 100 to 180 % power to determine its influence on the thermal-hydraulics parameters, especially the points of onset of nucleate boiling (ONB) and the onset of significant void (OSV) which is also referred to as point of net vapor generation (NVG). The exit coolant temperature, predicted coolant temperature at ONB, predicted coolant temperature at OSV, maximum heat flux, variation of linear heat generation rate along the heat channel, minimum point of critical heat flux (CHF) and the corresponding minimum departure from nucleate boiling ratio (MDNBR) were all noted for each power step. The result showed that at about 130 % power, the descending predicted coolant temperature at ONB crossed below the ascending exit coolant temperature, indicating an ONB, furthermore, at about 165 % power, the descending predicted coolant temperature at OSV crossed below the ascending exit coolant temperature, indicating an OSV. In this calculation, there were no variation in any other parameter of the reactor and the inlet enthalpy remained constant throughout the step variation of power.

1. Introduction

One of the most crucial parameters in nuclear reactor operation is heat flux. It describes how the heated fluid in a flow channel distributes heat per unit area. Evaluating the reactor core's point of critical heat flux is one of the biggest thermal-hydraulic problems. There are differences in the names given to CHF in Light Water Reactors (PWR and BWRs). In PWRs, it is called Departure from Nucleate Boiling (DNB) and it is a local phenomenon which varies from point to point in the channel, whereas in BWRs, it is called Dry-out and it is a global phenomenon, it encompasses all the channel as a whole. This is why the MDNBR depends on the channel heat flux, while the minimum critical power ratio (MCPR) depends on the channel thermal power. Whatever the prevailing mechanism, CHF may cause a sudden and significant temperature spike that could further cause an early burnout and a number of potential safety incidents if the working fluid's ability to remove heat from the heated cladding surface is somewhat compromised, if the rate of heat removal is lower than the rate of heat generation, or if coolant loss is the cause of the inability of the coolant to remove all the heat generated by the fuel element over a specific area within a given time frame [1]. In order to ensure reactor safety and nuclear fuel design, CHF is one of the most crucial phenomena to look into. Enhancing safety and improving operating economics are the outcomes of improving CHF prediction. Because of this, the nuclear industry must talk about and conduct research on CHF enhancement and forecast accuracy [1] In this work, we investigate the response of thermal hydraulics parameters to power excursion in VVER-1200, this is to ascertain the points of possible thermal crisis during this stepwise increase in power. It is worthy of note that in the design of VVER-1200 reactor, there are quite a number of design safety limits imposed on certain thermal-hydraulics parameters such as; pressure, coolant temperature, clad temperature, fuel centerline temperature, MDNBR etc. These safety limits help operators to carryout reactor operation safely, it also helps the structures systems and components (SSCs) important to safety to be able to serve their design lifetime without major problem.

2. Calculation method

We calculated the various thermal hydraulics parameters with the various formulas shown below.

The linear heat flux, q_l' is a measure of heat generated per unit length of the fuel rod. It is in direct proportionality to the area heat flux and volumetric heat flux.

$$q_l'(z)=q_{l,\max}' \cdot \cos\left(\frac{\pi \cdot z}{H_{eff}}\right). \tag{1}$$

Using the modified Zuber and Findlay [2] model neglecting the effect of drift velocity with a distribution parameter, C_o of 1.3, we calculated the area void fraction, α as follows where ρ_l and ρ_v are liquid and vapor densities:

$$\alpha=\frac{x_{true}\rho_l}{C_o\left[x_{true}\rho_l+\left(1-x_{true}\right)\rho_v\right]}. \tag{2}$$

Manon true quality, x_{true} formula [3] for OSV void fraction prediction accounts for a non-zero true quality at OSV which was zero for the Levy model. The expression is shown here as, where x_{eq} is the equilibrium quality;

$$x_{true}(z)=x_{eq}(z)+\left[x_{OSV}-x_{eq}(z_{OSV})\right]\exp\left(\frac{x_{eq}(z)}{x_{eq}(z_{OSV})}-1\right). \tag{3}$$

Due to the inability of the Manon model to predict void fraction at the ONB, Delhaye formula [3] is used to predict void fraction at ONB. The expression is shown here as, where ξ is the adjustment parameter to fit with the OSV void fraction;

$$x_{true}(z)=0.01\xi\left\{x_{eq}(z)-x_{eq}(z_{ONB})\left\{\tanh\left(\frac{x_{eq}(z)}{x_{eq}(z_{ONB})}-1\right)+1\right\}\right\} \tag{4}$$

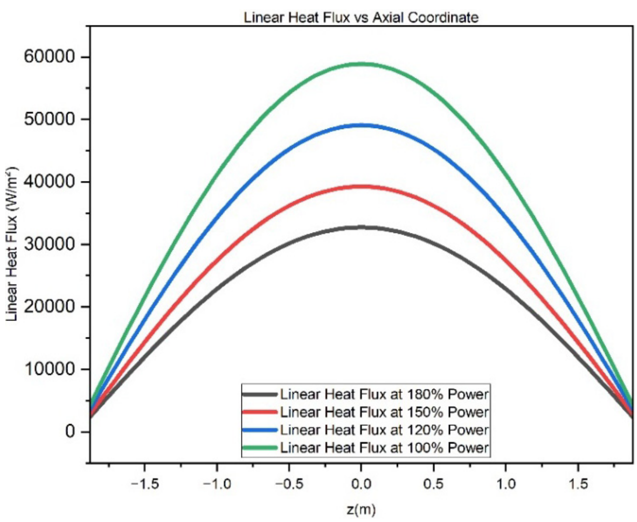
Using Saha and Zuber [4] and Frost and Dzakowic [5] relations, predicted coolant temperature, T_l at OSV and ONB respectively. T_{sat} is the saturation temperature, q'' is the heat flux, G is the mass flux, c_p is the specific heat capacity of the coolant, h_{fg} is the heat of vaporization, h_{lo} is the coolant heat transfer coefficient, σ is the coolant surface tension, k_l is the thermal conductivity of coolant, $\Pr_l$ is the coolant Prandtl number.

$$T_l\left(z_{OSV}\right)=T_{sat}-153.8\frac{q''}{Gc_p}, \tag{5}$$

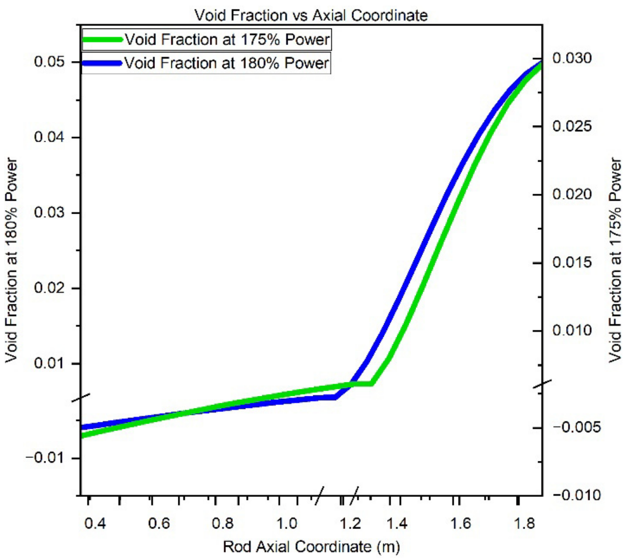
$$T_l\left(z_{ONB}\right)+\frac{q''}{h_{lo}}=T_{sat}+\left(\frac{8q''\sigma T_{sat}}{\rho_v h_{fg} k_l}\right)^{0.5}\Pr_l. \tag{6}$$

3. Results and Analysis

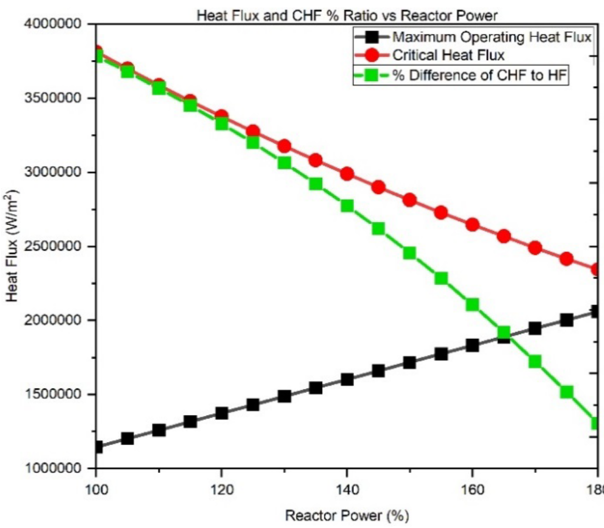
Calculation of thermal and hydraulic characteristics was carried out using initial data: thermal power of reactor, 3200MW$_{th}$; pressure of the coolant in the reactor, 16.2 MPa; and inlet temperature of the coolant, 298.2 °C. In the analysis of the result, the linear heat flux was calculated for four (4) different power step increase; at 100, 120, 150 and 180 % rated thermal power respectively. It was found that the linear heat flux increases in direct proportionality to the power increment as shown in Fig 1 (a), indicating a linear relationship between the thermal power and linear heat flux.



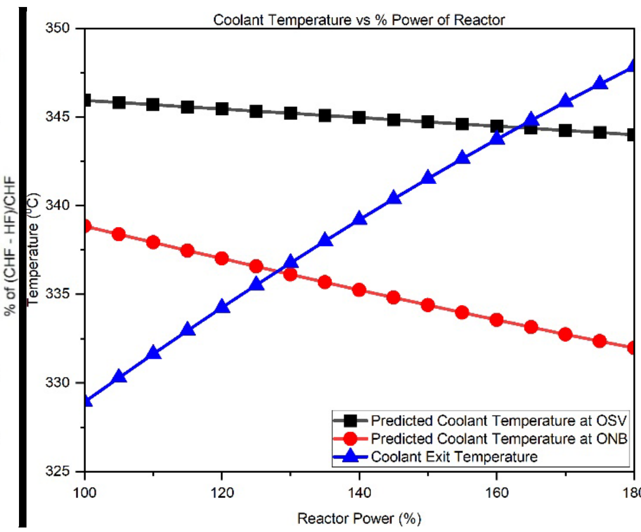
(a)



(b)



(c)



(d)

*Fig. 1. (a) Linear heat flux vs height, (b) Void fraction vs height,
(c) Heat flux density vs reactor power, (d) Coolant temperature vs reactor power*

The void fraction distribution in Fig1 (b) are for 175 and 180 % power, the OSV point actually started at 165 % power, but it was at these powers (175 and 180 %) that the boiling became a fully developed boiling encompassing both the ONB and OSV regions. As seen from the Fig 1 (b), the OSV point at 180 % occurred at about 3m of channel height, while that of 175 % power occurred at about 3.17 m. The OSV point also mark a major milestone in the heat removal in PWR reactor core, it indicates the beginning of thermal crisis which is the point of intersection of the maximum heat flux and CHF ratio as seen in Fig 1 (c). The maximum heat flux increased as the power increased while the critical heat flux continuously decreased as seen in Fig 1 (c). This means as the power increases the ratio of CHF to maximum heat flux will continue to decrease until it gets to a safety limit beyond which a thermal crisis will take place, this limit point is the point of intersection of the maximum heat flux and the CHF ratio shown in Fig 1 (c). The ascending exit coolant temperature and the descending predicted coolant temperature at ONB and OSV indicates that as the power increases, the exit coolant temperature will meet the temperature at ONB and at OSV. The point at which the ONB temperature meets the coolant temperature, is the ONB point, this happened at 130 % power, while the point at which OSV temperature meets the coolant temperature is the OSV point, this happened at 165 % power.

3. Conclusion

The conclusion of this calculation is that by theoretically increasing VVER-1200 thermal power beyond the maximum permissible thermal power of about 102–105 % power, while keeping other influencing parameters constant, the void fraction will increase, this increment will eventually lead to thermal crisis if the incremental power goes on continuously, as seen in Fig 1 (b). Also, it was noted in this calculation that the predicted CHF value will continue to race towards the maximum operating heat flux value until the CHF ratio intercepts the maximum heat flux which marks the beginning of thermal crisis, as seen in Fig 1 (d).

REFERENCES

1. Yang B.W. et al. Progress in rod bundle CHF in the past 40 years // Nuclear Engineering and Design. – 2021. – Iss. 376. – P. 111076.
2. Cai C. et al. Assessment of void fraction models and correlations for subcooled boiling in vertical upflow in a circular tube // International Journal of Heat and Mass Transfer. – 2021. – Iss. 171. – P. 121060.
3. Delhaye J.M., Maugin F., Ochterbeck J.M. Void fraction predictions in forced convective subcooled boiling of water between 10 and 18 MPa // International journal of heat and mass transfer. – 2004. – V. 47. – Iss. 19-20. – P. 4415–4425.
4. Pradip S., Zuber N. Point of net vapor generation and vapor void fraction in subcooled boiling // International Heat Transfer Conference: Digital Library. Begel House Inc., 1974.
5. Frost W., Dzakowic G.S. An extension of the method for predicting incipient boiling on commercially finished surfaces. USA: ASME, 1967.

СИСТЕМЫ ТРИГЕНЕРАЦИИ НА ОСНОВЕ ГАЗОВЫХ ТУРБИН

К.Д. Вьюгова

ФГБОУ ВО «КГЭУ», г. Казань, Республика Татарстан

Научный руководитель: А.Ш. Низамова, к-т техн. наук